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April 20, 2006

10 CFR 50.73
10 CFR 21

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

LaSalle County Station, Unit 1
Facility Operating License No. NPF 11
NRC Docket No. 50-373

Subject: Licensee Event Report

In accordance with 10 CFR 50.73 (a)(2)(iv)(A) and 10 CFR 21, Exelon Generation Company, (EGC), LLC, is submitting Licensee Event Report Number 06-001-00, Docket No. 050-373.

Should you have any questions concerning this letter, please contact Mr. Terrence W. Simpkin, Regulatory Assurance Manager, at (815) 415-2800.

Respectfully,

Daniel J Enright

Daniel Enright
Plant Manager
LaSalle County Station

Attachment: Licensee Event Report

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - LaSalle County Station

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE Three Control Rods Fail to Indicate Fully Inserted Following Automatic Reactor Scram

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	20	2006	2006	- 001 -	00	04	20	2006		DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	006

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

☐ 20.2201(b)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2201(d)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(1)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☒ OTHER
☐ 20.2203(a)(2)(iii)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	Specify in Abstract below or in NRC From 366A
☐ 20.2203(a)(2)(iv)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(viii)(A)	
☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Larry Coyle, Site Operations Director	815-415-2200

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	EK	XPT	B093	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE)	☒ NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)

On 2/20/06, Unit 1 was at approximately six percent power while shutting down for a scheduled refueling outage. At 0023 hours, approximately 20 minutes after the main turbine was taken offline, all of the main turbine bypass valves unexpectedly opened. The resulting drop in reactor and steam line pressure caused a Group 1 Primary Containment isolation and an automatic reactor scram.

Following the scram, the operators observed that three control rods did not indicate fully inserted. Due to the inability to verify that shutdown criteria were met, a Site Area Emergency (SAE) was declared at 0028 hours in accordance with the current Emergency Action Level guidance. The SAE was exited and the site entered Recovery mode at 0427 hours, following determination that the reactor was in fact fully shutdown.

The cause of the scram was a failed power supply in the main turbine EHC system. The three control rods failed to fully insert to position 00 due to friction from channel distortion. The EHC power supply was replaced, and actions were taken to address the channel distortion issue, including discharging fuel assemblies susceptible to channel bowing in the next operating cycle or placing them in non-control cell locations.

17. NARRATIVE (If more space is required, use additional copies of NRC Form 366A)

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor, 3489 Megawatts Thermal Rated Core Power

A. CONDITION PRIOR TO EVENT

Unit(s): 1 Event Date: 02/20/06 Event Time: 0023 CST
Reactor Mode(s): 1 Power Level(s): 006
Mode(s) Name: Run

B. DESCRIPTION OF EVENT

On 2/20/06, Unit 1 was shutting down for scheduled refueling outage. At 0023 hours, approximately 20 minutes after the main turbine was taken offline and with the reactor at six percent power, all five of the turbine bypass valves unexpectedly opened. The resulting drop in reactor and steam line pressure caused a Group 1 Primary Containment isolation and an automatic reactor scram.

Following the scram, the control room operators observed that three control rods (38-43, 26-15 and 34-47) failed to indicate full in. With the position of three control rods indeterminate, the operators entered LGA-010, "Failure to Scram" and, because they could not verify that shutdown criteria were met in accordance with Emergency Action Level (EAL) MS-3, they declared a Site Area Emergency (SAE) at 0028 hours.

Following attempts to manually insert the rods, the operators re-initialized the Rod Worth Minimizer (RWM). After re-initialization, the RWM indicated that all rods were full in. LGA-010 was exited and the scram was reset. However, once the scram was reset, the RWM again indicated that the same three control rods were not full in. LGA-010 was entered again. At 0047 hours the Alternate Rod Insertion (ARI) system was actuated, and RWM "A" indicated that all rods were fully in the core. Following verification that the reactor was fully shutdown, the SAE was exited and the site entered Recovery mode at 0427 hours on 2/20/06.

This event was determined to be reportable in accordance with 10 CFR 50.72(b)(3)(iv) as a valid actuation of the Reactor Protection System while the reactor was critical, and in accordance with 10 CFR 50.72(a)(1)(i) as a declaration of one of the emergency classes in the Station's Emergency Plan. ENS notification #42348 was made at 0114 hours.

In addition, this 10 CFR 50.73 report satisfies the 10 CFR 21 requirements for reporting of defects concerning channel distortion that were identified in this event.

C. CAUSE OF EVENT

The scram occurred due to the unexpected opening of the turbine bypass valves. The bypass valves opened due to a perturbation in the electro-hydraulic control (EHC) system control circuitry caused by a voltage dip on the -22 VDC bus feeding the analog EHC circuit cards. The voltage dip was caused by an intermittent failure of the -22 VDC power supply. When the -22 VDC power supply failed, the -22 VDC bus was transferred to the -22 VDC permanent magnet generator (PMG) supply; however, because the main turbine was coasting down, the PMG was operating at only one-half of the nominal voltage.

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The voltage transient changed the pressure setpoint (along with several other parameters), causing the bypass valves to open. The root cause of the -22 VDC power supply failure could not be determined.

The cause of the failure of control rods 26-15, 34-47 and 38-43 to fully insert was that the scram pressure at low power (864 psig) did not provide enough force to overcome friction due to fuel channel distortion that stopped the control rods prior to the end of travel. Based on the sequence of events and the available indications and analyses, it has been determined that control rods 26-15 and 34-47 inserted to a position between notches 00 and 01, which is fully inserted into the core. Control rod 38-43 inserted to a position between notches 06 and 16, with the best data fit near notch 12. Because all three rods stopped in a position between notches, the Rod Position Indication System was unable to indicate their position.

The fuel cells associated with control cells 26-15, 38-43 and 34-47 were removed and visually inspected. The blade associated with cell 38-43 was also removed and inspected. Two of the four fuel bundles adjacent to control rod 38-43 had visible channel distortion. In addition, indications of control rod blade to fuel channel contact on the Areva fuel and the control blades for control cell 26-15 and 34-47 were also observed. Rub marks were observed on the control rod blade consistent with channel interaction.

Industry experience of BWR/6 and BWR/4 & 5 C-lattice plants with GNF thick/thin channels indicates an increasing occurrence of additional control blade friction that has been manifested as degraded control rod operational performance. Investigations indicate the cause to be control rod blade-to-channel interference due to an unexpected amount of channel distortion. Although there are many contributors to channel distortion, the three primary sources of interest for this report have been identified as: (1) fast fluence gradient-induced channel bow, (2) channel bulge, and (3) control blade shadow corrosion-induced channel bow. These phenomena progress rather slowly and manifest themselves in operational problems at high exposure.

GE has been working with the industry on these issues since the 1970s. GE issued SIL 320, "Mitigation of the Effects of Peripheral Core Location on Fuel Channel Bowing" in December 1979, with supplements in February 1984, July 1998, and April 2004. GE also issued 10 CFR Part 21 Communication SC05-06 in July 2005. This information included criteria for determining which channels are susceptible to distortion, and surveillance recommendations to mitigate fuel channel distortion and its effects.

Procedure LOS-RD-SR7, "Channel Interference Monitoring" provides specific guidance based upon the GE Part 21 Communication, for testing control rods using full insertion timing. LOS-RD-SR7 is in compliance with the GE Part 21 communication; however, the procedure does not include optional guidance from GE suggesting a 125 percent limit on scram time deviation from the core average. Control rods 26-15, 38-43 and 34-47 were all identified as susceptible and had been tested in accordance with LOS-RD-SR7. Rod 38-43 had indicated a failure to settle and an increasing scram time, but was not declared inoperable.

Control rod position assessment following the scram was complicated by the re-initialization of the RWM. During the investigation, it was found that the RWM computer code contains a critical flaw in that the re-initializing routine writes "00" for the rod positions, and the scram capture mode locks this value in. As a

17. NARRATIVE (If more space is required, use additional copies of NRC Form 366A)

result, the RWM system provided incorrect information to the operators when RWM was reinitialized with the scram capture mode still active.

Finally, the declaration of the SAE was appropriate given current procedural guidance. However, given that the reactor was, in fact, fully shutdown immediately following the scram, the SAE was excessively conservative and resulted in unnecessarily exercising government and company emergency response assets. The root cause of this problem was that EAL MS-3 does not provide the operational flexibility to evaluate plant conditions and use alternate indications, such as nuclear instrumentation, to verify that the reactor is shutdown.

D. SAFETY ANALYSIS

The safety significance of this event was low. Although the event is classified as an unplanned scram with a loss of normal heat removal, the scram occurred from low power during a planned shutdown. The heat removal capabilities of the feedwater system and the safety relief valves were adequate, and ECCS was not challenged.

Following the scram, all control rods, with the exception of rod 38-43, inserted fully into the core; i.e., to a position of 02 or beyond. LaSalle core analyses show that the core will remain shutdown under all conditions with all rods except one inserted to 02 or beyond. In addition, a specific post-event evaluation determined that the core would have remained shutdown under cold, xenon-free conditions even if control rods 26-15, 38-43 and 34-47 had been full out.

E. CORRECTIVE ACTIONS

1. Both the normal and PMG -22 VDC power supplies were replaced on Unit 1 (Complete).
2. Monitoring capability was installed for the -22 VDC power supplies (Complete).
3. The current analog EHC system will be replaced on both Units with a more single failure-tolerant digital EHC system (AT# 456319-13, 14).
4. Prior to startup from the Unit 1 refueling outage, the core was reconfigured to either discharge the Atrium-10 fuel assemblies that would have been susceptible to channel bowing in the next operating cycle or place them in non-control cell locations (Complete).
5. Fourteen potentially susceptible GNF GE-14 fuel assemblies were re-channeled prior to reinstallation (Complete).
6. The criterion for determining potential susceptibility to channel distortion was expanded to provide for testing fuel assembles at the 80% susceptibility point. This is intended to add margin to ensure testing occurs before fuel channel friction can become an issue (Complete).
7. Procedure LOS-RD-SR7 was revised to: 1) incorporate criteria for declaring a rod inoperable based on scram time performance relative to average core scram time; 2) change the testing methodology to take into account changes in stall flow during the testing sequence; and 3) to require trending of settle times with appropriate management notification required for adverse trends (Complete).

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8. Unit 2 will be evaluated and mitigated through continued surveillance activities utilizing enhanced testing methodologies to detect channel distortion.

9. An interim enhancement to Anticipated Transient Without Scram-related EALs MA3 and MS3 was implemented to clarify and improve the criteria and definition of shutdown conditions, as allowed by 10CFR50.54(q) (Complete).

10. NRC approval will be obtained to implement plant specific shutdown criteria in the M_3 series LaSalle EALs as an alternative method per RIS 2003-18, Supplement 2 (AT# 456768-16, 17).

F. PREVIOUS OCCURRENCES

A search of Licensee Event Reports going back 10 years found no previous occurrences at LaSalle of a reactor scram due to an EHC power supply failure, or of control rods not fully inserting following a scram due to friction from channel distortion.

G. COMPONENT FAILURE DATA

Lamda Electronics, -22VDC House Power Supply, Model# LMF-24-OVMY-R-3396-3